RELATED CORRESPONDENCE

LIC 10/28/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC
OCT 30 1980
Office of the Secretary
Docketing & Service
Branch

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
	)	
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

LICENSEE'S TESTIMONY OF

ROBERT C. JONES, JR. AND T. GARY BROUGHTON

IN RESPONSE TO THE BOARD QUESTION ON UCS CONTENTION 8

8011060404

OUTLINE

This testimony supplements Licensee's Testimony of Robert C. Jones, Jr., and T. Gary Broughton in Response to UCS Contention No. 8 and ECNP Contention No. 1(e) (Additional LOCA Analysis), dated September 15, 1980. In particular, this testimony responds to the one aspect of the Board Question on UCS Contention 8 which was not addressed by the earlier testimony -- namely, the recommendations made in NUREG-0565 and NUREG-0623.

INDEX

Page

<u>INTRODUCTION</u>

This testimony, by Mr. Robert C. Jones, Jr., Supervisory Engineer, ECCS Analysis Unit, Babcock & Wilcox Company, and Mr. T. Gary Broughton, GPU Control and Safety Analysis Manager, is addressed to the following Board Question regarding UCS Contention 8:

BOARD QUESTION REGARDING UCS CONTENTION 8

> The board directs the staff and the licensee to present experts and the fundamental documents involved in the small break LOCA analysis, and to have very complete testimony on this subject. The recommendations of NUREG-0565 and NUREG-0623 should be addressed.
>
> It appears from the small break LOCA analysis that there is a large amount of reliance upon operator action and on non-safety grade equipment. The board wants that issue explored by testimony, including why such reliance is proper.

<u>RESPONSE</u>

<u>BY WITNESSES JONES AND BROUGHTON</u>:

Licensee's testimony in response to UCS Contention 8 addresses the small break loss of coolant accident (LOCA) analyses which have been performed to support the operation of TMI-1. The exhibits identified as items 3-13 in Licensee's Certificate of Service dated September 15, 1980, and provided to the parties pursuant thereto, present the fundamental results of these small break LOCA analyses.

The limited extent to which operator action and non-safety-grade equipment are utilized in the analyses for accident mitigation is discussed in the previously filed testimony (pages 3, 4, 8 and 9). Those discussions also address why such reliance is appropriate.

The following is a response to each of the recommendations (applicable to licensees) presented in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants," and in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors."

NUREG-0565, RECOMMENDATION 2.1.2.a

> Provide a system which will assure that the block valve protects against a stuck-open PORV. This system will cause the block valve to close when RCS pressure has decreased to some value below the pressure at which the PORV should have reseated. This system should incorporate an override feature. Each licensee should perform a confirmatory test of the automatic block valve closure system.

RESPONSE

BY WITNESS BROUGHTON:

Design and installation of an automatic PORV block valve closure system is not being pursued at this time. The need for such a system has not been determined by appropriate analysis, which is called for by Item II.K.3.7 of NUREG-0660. Furthermore, it is not obvious that the addition of a closure system would be a modification which would provide greater safety, since the system may result in an increased probability of challenge to the pressurizer safety valves. Until the evaluations in response to Item II.K.3.7 are completed, the need to design and install an automatic block valve closure system has not been established.

NUREG-0565, RECOMMENDATION 2.1.2.b

> Most overpressure transients should not result in the PORV opening. Therefore, licensees should document that the PORV will open in less than five percent of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

RESPONSE

BY WITNESS JONES:

Anticipated transients which produce an increase in Reactor Coolant System (RCS) pressure and which might cause the pressurizer power operated relief valve (PORV) to open include loss of feedwater, loss of external electrical load, turbine trip, uncontrolled control rod withdrawal from startup conditions, inadvertent closure of main steam isolation valves (MSIV's), and inadvertent moderator boron dilution. For any of these events the greatest potential for opening the PORV exists at the beginning of the fuel cycle when there is the minimum beneficial reactivity feedback. As the fuel cycle progresses, the moderator and Doppler negative reactivity feedback increases, thereby diminishing the magnitude of overpressurization. Also, as shown below, not every overpressurization event results in opening the PORV.

Overpressurization due to a loss of main feedwater, loss of electrical load or turbine trip will not cause the PORV to open because of the anticipatory trip functions which have been installed at TMI-1 and because of the increased PORV opening set pressure. This is the case at any time in the fuel cycle.

Safety analyses performed for TMI-1 (Final Safety Analysis Report) of the moderator dilution event at full power indicate peak system pressures lower than the present PORV opening setpoint. The lowered high pressure trip setpoint provides further assurance that the PORV will not open.

Inadvertent closure of the MSIV's does not result in a direct reactor trip and will result in an increase in primary system pressure. The most severe results from this event would involve closure of all MSIV's in a short time (a few seconds). At TMI-1, however, the MSIV closure time is about 2 minutes and inadvertent closure of the MSIV's is not expected to result in PORV actuation. Also, no inadvertent closure of all MSIV's has been experienced on a B&W plant to date.

Inadvertent control rod withdrawal from startup conditions can result in primary system overpressurization for a narrow range of small reactivity insertion rates. These are events which result in a relatively slow overpressurization requiring actuation of the high reactor coolant pressure trip rather than a high flux trip. The lowered high pressure trip setpoint and increased PORV opening setpoint, however, reduce the potential for PORV opening. Also, an event of this nature has not happened at a B&W plant to date.

In summary, there are some overpressurization events which can lead to PORV opening. Anticipated transients which have occurred, however, will not now result in PORV actuation due to the addition of anticipatory trip functions and the revision of the high pressure trip and PORV opening set points. Other, less frequent events which can currently result in PORV opening have not occurred to date at a B&W plant. Therefore, while no quantitative assessment of PORV opening has been performed for overpressurization events, it is readily apparent that this fraction is less than 5%.

NUREG-0565, RECOMMENDATION 2.1.2.c

All failures of PORVs to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

NUREG-0565, RECOMMENDATION 2.1.2.e

All failures of safety valves to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

RESPONSE

BY WITNESS BROUGHTON:

Licensee will propose changes to the TMI-1 Technical Specifications that will require reporting of failures or challenges to the PORV and safety valves as recommended.

NUREG-0565, RECOMMENDATION 2.1.2.d

Licensees should submit a report to the NRC which discusses the safety valve failure rate experienced in B&W operating plants.

RESPONSE

BY WITNESS BROUGHTON:

Licensee is unaware of any instances of failures of Reactor Coolant System safety valves at any B&W plant. See Licensee's testimony in response to the Board Question on UCS Contention 6.

NUREG-0565, RECOMMENDATION 2.2.2.a

> The analysis methods used for small break LOCA analysis by B&W should be revised, documented, and submitted for NRC approval.

NUREG-0565, RECOMMENDATION 2.2.2.b

> Plant-specific calculations using the NRC approved model for small breaks should be submitted by all licensees to show compliance with 10 CFR 50.46.

RESPONSE

BY WITNESS JONES:

The small break LOCA analyses which were performed after the TMI-2 accident were done to provide an improved analytical basis for emergency procedures for small break LOCA's. These analyses were not performed to demonstrate compliance with 10 CFR 50.46. NUREG-0565 states that the post-TMI-2 analyses are beyond those normally considered in small break analyses and that the NRC Staff has some concerns relative to the use of the currently approved small break model for these purposes. However, NUREG-0565 (Section 2.2.I) also contains the following conclusion: "The small break analysis methods used by B&W are satisfactory for the purpose of predicting trends in plant behavior following small break LOCAs and for training of reactor operators." NUREG-0565 does not state that the approved B&W small break evaluation is difficient for demonstrating compliance for TMI-1 with respect to 10 CFR 50.46 and Appendix K. While further code development may be performed and model modifications may be made, the changes are not expected to result in a substantial change to the Appendix K evaluations performed for TMI-1.

NUREG-0565, RECOMMENDATION 2.2.2.c

> The effects of core flood tank injection on small break LOCAs should be further investigated to determine the amount of condensation realistically expected and to determine its effect on heatup and core uncovering. The condensation model and modeling procedures (i.e., injection location used in the computer analyses) require further investigation to assure that the effects of CFT injection are biased in a conservative manner. Semiscale and LOFT test data should be used to verify the models.

RESPONSE

BY WITNESS JONES:

This Staff concern relates to the potential for a large underprediction of system pressure, due to the analytical assumption of instantaneous steam condensation on the cold Core Flood Tank (CFT) water delivered to the RCS during a small break. Contrary to this concern, the small break analyses performed for TMI-1 do not predict large pressure oscillations caused by core flood injection. Thus, while further examination of this phenomena may be performed, the small break predictions are not expected to be substantially altered.

NUREG-0565, RECOMMENDATION 2.3.2.a

Tripping of the RCPs in the event of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break problem, for example, an increase in the HPI flow rate. In the interim, until a better solution is found, the RCPs should be tripped automatically in the case of a small break LOCA. The signals designated to initiate the RCP trip should be carefully selected in order to differentiate between a small break LOCA and other events which do not require the RCPs to be tripped.

NUREG-0623, CONCLUSION 6.0(4)

From items (2) and (3), above, we find that tripping all of the reactor coolant pumps during small break LOCAs is required at this time, and that this pump trip should be automatically initiated from equipment that is safety-grade to the extent possible.

NUREG-0623, CONCLUSION 6.0(5)

The impact of an early pump trip on non-LOCA transients is not predicted to lead to unacceptable consequences. However, tripping the reactor coolant pumps for non-LOCA transients can aggravate the consequences of these transients and extend the time required to bring the plant into controlled shutdown condition. For B&W plants, tripping of the reactor coolant pumps during severe overcooling events increases the potential for interruption of natural circulation due to steam formation in the coolant loops.

Therefore, we conclude that the criteria and requirements for reactor coolant pump trip to be established from item (4), above, should minimize, to the extent practicable, the probability of initiating a reactor coolant pump trip for non-LOCA transients.

NUREG-0623, CONCLUSION 6.0(6)

The staff recognizes the potential desirability of running the reactor coolant pumps to provide forced circulation during small break LOCAs and we encourage the continued exploration by the industry of means by which this could be accomplished. For example, an increase in HPI capacity or two-pump operation as proposed by Combustion Engineering are a step in this direction.

RESPONSE

BY WITNESS BROUGHTON:

The TMI-1 Restart Report, Supplement 1, Part 3, response to question 11, presents the design characteristics of our proposed reactor coolant pump trip system. This system is based on a coincident loss of sub-cooling margin and high pressure injection actuation. The NRC Staff has accepted this approach as described in NUREG-0680 (SER at p. C2-18).

NUREG-0565, RECOMMENDATION 2.3.2.b

> The B&W small break LOCA analyses rely on equipment which has not previously been characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to provide the necessary RCP trip, the pressurizer PORV and PORV block valve, and equipment used to actuate the PORV and PORV block valve fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to comply with the requirement of Section 9 of NUREG-0585, regarding the interaction of non-safety and safety-grade system.

RESPONSE

BY WITNESS JONES:

The equipment used in the post TMI-2 accident small break LOCA analyses (the analyses addressed in NUREG-0565) which is not part of the Reactor Protection System or part of the engineered safety features is identified in Licensee's testimony in response to UCS Contention 8 and ECNP Contention 1 (Additional LOCA Analysis) (pages 3, 4, 8 and 9).

The specific items utilized in the analyses are the Emergency Feedwater System and the equipment used to provide reactor coolant pump trip. The pressurizer power operated relief valve (PORV) and PORV block valve have not been relied upon in the LOCA analyses.

NUREG-0565, RECOMMENDATION 2.3.2.c

Plant simulators used for operator training should offer, as a minimum, the following small break LOCA events:

(1) continuous depressurization;

(2) pressure stabilized at a value close to secondary system pressure;

(3) [illegible]ressurization;

(4) stuck-open PORV; and

(5) stuck-open letdown valve.

Each of these cases should be simulated with RCPs running as well as tripped. The first three events should be simulated for both cold and hot leg breaks. In addition to the usual assumed single failures in the ECCS and feedwater systems, complete loss of feedwater should also be simulated in conjunction with the above events. It is important that training programs also expose the operators to various kinds of system transients on inadequate core cooling as discussed in Section 2.1.9 of NUREG-0578.

RESPONSE

BY WITNESS BROUGHTON:

Operator training, including the use of simulators, will be addressed in Licensee's testimony on management competence.

NUREG-0565, RECOMMENDATION 2.6.2.a

> The various modes of two-phase natural circulation, which are expected to play a significant role in plant response following a small break LOCA, should be demonstrated experimentally. In addition, the staff requires that the licensees provide verification of their analysis models to predict two-phase natural circulation by comparison of the analytical model results to appropriate integral systems tests.

RESPONSE

BY WITNESS JONES:

The B&W small break LOCA evaluation model includes appropriate consideration of the mechanisms responsible for natural circulation. The computer code utilized models both density changes and flow losses under single- and two-phase fluid conditions. Thus, the evaluation model should reasonably predict the various modes of two-phase natural circulation. Additionally, for small break LOCA's, the steam generators do not have an important influence on the transient except for those cases where the break size is insufficient to discharge energy at least equal to that added by the core decay heat. As noted in Licensee's testimony in response to UCS Contentions 1 and 2 (Natural and Forced Circulation) (pages 6 and 7), this break size would be less than approximately 0.02 ft^2. Breaks smaller than 0.02 ft^2 will retain substantially more system inventory than the design basis small break, which is approximately 0.07 ft^2, and have large margins relative to the potential for core uncovery. Therefore, while further examination of two-phase natural circulation phenomena may be performed, TMI-1 is still expected to conform to 10 CFR 50.46.

NUREG-0565, RECOMMENDATION 2.6.2.b

> Appropriate means, including additional instrumentation, if necessary, should be provided in the control room to facilitate checking whether natural circulation has been established.

RESPONSE

BY WITNESS BROUGHTON:

Checks that natural circulation has been established are included in appropriate plant procedures and require observing primary system hot and cold leg temperatures for a constant differential and observing that cold leg temperature approaches secondary system saturation temperature. The instrumentation used in this determination are located in the control room.

NUREG-0565, RECOMMENDATION 2.6.2.c

Licensees should provide an analysis which shows the plant response to a small break which is isolated and the PORV fails-open upon repressurization of the reactor coolant system to the PORV setpoint.

RESPONSE

BY WITNESS JONES:

A specific analysis providing the plant response to a small break which is isolated and the PORV fails-open upon repressurization of the RCS to the PORV setpoint has not been performed. However, based on the analyses discussed in Licensee's testimony in response to UCS Contention 8 and ECNP Contention 1(e) (Additional LOCA Analysis), the response to this event can be described.

Initially, as a result of the small break, the system will depressurize. Actuation of the High Pressure Injection system (HPI) will automatically occur, assuming feedwater availability, prior to the loss of natural circulation. Should break isolation occur after natural circulation is lost and prior to the establishment of the boiler-condenser mode of steam generator heat removal, system repressurization would occur. Assuming that the repressurization reaches the PORV setpoint and that the PORV subsequently sticks open, a transient very similar to that calculated for a PORV initially stuck open would then occur. Adequate core cooling would be continuously maintained for this transient by the fluid provided by HPI.

NUREG-0565, RECOMMENDATION 2.6.2.d

> Licensees should provide an analysis which shows the plant response to a small break in the pressurizer spray line with a failure of the spray isolation valve to close.

RESPONSE

BY WITNESS JONES:

A break in the pressurizer spray line along with a failure of the spray isolation valve to close results in inventory loss from both the RCS cold leg and the top of the pressurizer. The leak rates from the cold leg would be limited by the area of the spray line, 0.025 ft^2, and from the pressurizer the leak rate would be limited by the flow area of the spray nozzle in the pressurizer, 0.072 ft^2. The small break LOCA analyses performed for TMI-1 to demonstrate conformance to 10 CFR 50.46 envelope the total leak flow area for this case. Thus, system inventory losses similar to that which would occur for this scenario have already been considered in the LOCA analyses. However, for this accident, liquid inventory would remain in the pressurizer while the TMI-1 small break analyses empty the pressurizer. The effect of the stored inventory in the pressurizer for this event is expected to be offset by the increased availability of HPI for core cooling. In the analyses performed for TMI-1, less than 70% of the HPI was calculated to enter the core due to the direct bypass of the injected fluid out the break, which was assumed to be located in the bottom of the cold leg pump discharge piping between the HPI nozzle and the reactor vessel. For the spray line break, no HPI fluid would bypass out the break without first entering the vessel. The increased HPI flow for the spray line break would establish long term cooling earlier, relative to an equivalently sized pump discharge break, and is expected to offset the effect of the stored inventory in the pressurizer. Therefore, an analysis of this accident is not expected to provide results which are in excess of 10 CFR 50.46 limits.

NUREG-0565, RECOMMENDATION 2.6.2.e

> Licensees should provide confirmatory information to show that HPI and CFT flows during small breaks are insufficient to form water slugs, or if they do, to show that the structural design bases of the primary system includes loads due to:
>
> 1. water slug intertial motion;
> 2. water slug impact; and
> 3. pressure oscillation due to steam condensation

RESPONSE

BY WITNESS JONES:

During small breaks, water slugs are not expected to be formed as a result of HPI and CFT flows. The HPI flows would be less than 140 ft^3/min during a small break transient. Since the piping volume from the HPI nozzle to the reactor vessel is 280 ft^3, it would take two minutes to fill the pipe. Also, the reactor vessel internals vent valves will continuously equalize pressures throughout the primary system. Therefore, the HPI water will drain into the vessel and there is no mechanism available to hold the HPI water in the cold leg pipe. Thus, slug flow as a result of the HPI will not occur.

The water injected from the CFT's also is not expected to produce slug flow since the fluid is directly injected into the reactor vessel downcomer. Also, the internals vent valves minimize pressure gradients within the vessel such that no holdup of injected CFT water will occur. Thus, no water slugs will occur as a result of CFT injection.

NUREG-0565, RECOMMENDATION 2.6.2.f

> Licensees should provide an analysis of the possibility and impact of RCP seal damage and leakage due to loss of seal cooling on loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small break LOCA with subsequent RCP seal failure.

RESPONSE

BY WITNESS BROUGHTON:

This recommendation was addressed in Licensee's response to R.W. Reid's letter of November 21, 1979, which was provided by letter No. TLL-285, dated June 30, 1980. In this response, a description of the RCP seal system and its cooling was provided along with a discussion of the probable degradation mechanism, the time and methods available to restore seal cooling, and the result of loss of cooling for up to 60 minutes. The results of that analysis did not indicate that excessive seal leakage would occur within 60 minutes.

NUREG-0565. RECOMMENDATION 2.6.2.g

Licensees shall provide pretest predictions of LOFT Test L3-6 (Reactor Coolant Pumps Running).

NUREG-0623, CONCLUSION 6.0(7)

We will require verification of small break models with the pumps running against appropriate integral systems experimental tests. In particular, we will require that the PWR vendors and fuel suppliers perform pretest predictions of the LOFT SBLOCA test with pumps running scheduled to be performed in March of 1980.

RESPONSE

BY WITNESS BROUGHTON:

GPU is a participant in the B&W owners' group program to predict LOFT L3-6. This analysis will be performed by B&W and provided to the NRC.

NUREG-0565, RECOMMENDATION 2.6.2.h

With regard to the effects of noncondensible gases during a small break LOCA, the licensees should provide the following information:

1. The technical justification for omitting the radiolytic decomposition of injected ECC water as a source of noncondensible gas; and

2. Confirmatory information to verify the predicted condensation heat transfer degradation in the presence of noncondensible gases.

RESPONSE

BY WITNESS JONES:

Analyses of the effect of noncondensibles on the condensation heat transfer process in the steam generator during a small break LOCA have been performed. These analyses, which included the effects of radiolytic decomposition, determined that sufficient condensation surface would remain within the steam generator and that the boiler-condenser mode would not be prohibited. Additionally, even under a postulated condition that the noncondensible gases prohibited condensation, HPI can be operated in a feed and bleed mode to supply adequate core cooling - see Licensee's testimony in response to UCS Contentions 1 and 2 (Natural and Forced Circulation). Thus, while further examination of the effect of noncondensibles on the condensing heat transfer process within the steam generator may be performed, provisions are available at TMI-1 to assure adequate core cooling.

NUREG-0565, RECOMMENDATION 2.6.2.i

By use of analysis and/or experiment, address the mechanical effects of induced slug flow on steam generator tubes.

RESPONSE

BY WITNESS JONES:

Analysis of the effect of induced slug flow on the steam generator has been performed. The analysis assumed that a sudden front of water impacted the tube sheet with a flow equivalent to that of normal operation. It was assumed that this load was suddenly applied and that the entire load was absorbed by the tubes directly under the inlet nozzle of the steam generator. The loading on a steam generator tube was calculated to be 21.5 lbf, in comparison to the theoretical buckling load of approximately 700 lbf. Thus, induced slug flow will not affect the integrity of the steam generator tubes.

T. GARY BROUGHTON

Business Address:	GPU Service Corporation 100 Interpace Parkway Parsippany, New Jersey 07054
Education:	B.A., Mathematics, Dartmouth College, 1966.
Experience:	Control and Safety Analysis Manager, GPU Service Corporation, 1978 to present. Responsible for nuclear safety analysis and integrated thermal, hydraulic and control system analysis of nuclear and fossil plants. Supervised on-site technical support groups at Three Mile Island, Unit 2 during the post-accident period.
	Safety and Licensing Engineer; Safety and Licensing Manager, GPU Service Corporation, 1976 to 1978. Performed and supervised nuclear licensing, environmental licensing and safety analysis for Oyster Creek, Three Mile Island and Forked River plants. Served as Technical Secretary to Oyster Creek and Three Mile Island General Office Review Boards.
	Officer, U.S. Navy, 1966 to 1976. Trained at Naval Nuclear Power School, Prototype and Submarine School. Positions held include Nuclear Propulsion Plant Watch Supervisor, Instructor at D1G prototype plant and Engineering Officer aboard a fast-attack nuclear submarine.
Publications:	EPRI CCM-5, RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analyses of Complex Fluid Flow Systems, Volume 4: Applications, December, 1978, Section 6.1, "Analysis of Rapid Cooldown Transient - Three Mile Island Unit 2", with N.G. Trikouros and J. F. Harrison.

"The Use of RETRAN to Evaluate Alternate Accident Scenarios at TMI-2", with N. G. Trikouros. Proceedings of the ANS/ENS Topical Meeting on Thermal Reactor Safety, April 1980, CONF-800403.

"A Real-Time Method for Analyzing Nuclear Power Plant Transients", with P.S. Walsh. ANS Transactions, Volume 34 TANSAD 34 1-899 (1980).

ROBERT C. JONES, JR. POOR ORIGINAL

Business Address: Babcock & Wilcox Company
Nuclear Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

Education: B.S., Nuclear Engineering, Pennsylvania State University, 1971. Post Graduate Courses in Physics, Lynchburg College.

Experience: June 1971-June 1975: Engineer, ECCS Analysis Unit, B&W. Performed both large and small break ECCS analyses under both the Interim Acceptance Criteria and the present Acceptance Criteria of 10 CFR 50.46 and Appendix K.

June 1975-Present: Acting Supervisory Engineer and Supervisory Engineer, ECCS Analysis Unit, B&W. Responsible for calculation of large and small break ECCS evaluations, evaluations of mass and energy releases to the containment during a LOCA, and performance of best estimate pretest predictions of LOCA experiments as part of the NRC Standard Problem Program. Involved in the preparation of operator guidelines for small-break LOCA's and inadequate core cooling mitigation.

RELATED CORRESPONDENCE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC
OCT 30 1980
Office of the Secretary
Docketing & Service
Branch

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Testimony of Robert C. Jones, Jr. in Response to Board Questions 6.e and 6.f," "Licensee's Testimony of James H. Correa and Gary T. Urquhart in Response to the Board Question on UCS Contention 6" and "Licensee's Testimony of Robert C. Jones, Jr. and T. Gary Broughton in Response to the Board Question on UCS Contention 8" dated October 28, 1980 were served upon the following by hand delivery on October 28, 1980 and on those identified below with an asterisk by deposit in the U.S. mail, first class, postage prepaid, this 28th day of October, 1980.

Thomas A. Baxter

Thomas A. Baxter

POOR ORIGINAL

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

SERVICE LIST

Ivan W. Smith, Esquire
Chairman
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Walter H. Jordan
Atomic Safety and Licensing
Board Panel
881 West Outer Drive
Oak Ridge, Tennessee 37830

Dr. Linda W. Little
Atomic Safety and Licensing
Board Panel
5000 Hermitage Drive
Raleigh, North Carolina 27612

James R. Tourtellotte, Esquire
Office of the Executive Legal Director
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

* Docketing and Service Section
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

* John A. Levin, Esquire
Assistant Counsel
Pennsylvania Public Utility Comm'n
Post Office Box 3265
Harrisburg, Pennsylvania 17120

Karin W. Carter, Esquire
Assistant Attorney General
505 Executive House
Post Office Box 2357
Harrisburg, Pennsylvania 17120

* John E. Minnich
Chairman, Dauphin County Board
of Commissioners
Dauphin County Courthouse
Front and Market Streets
Harrisburg, Pennsylvania 17101

* Walter W. Cohen, Esquire
Consumer Advocate
Office of Consumer Advocate
1425 Strawberry Square
Harrisburg, Pennsylvania 17127

POOR ORIGINAL

* Jordan D. Cunningham, Esquire
Attorney for Newberry Township
T.M.I. Steering Committee
2320 North Second Street
Harrisburg, Pennsylvania 17110

Theodore A. Adler, Esquire
Widoff Reager Selkowitz & Adler
Post Office Box 1547
Harrisburg, Pennsylvania 17105

Ellyn R. Weiss, Esquire
Attorney for the Union of Concerned Scientists
Harmon & Weiss
1725 Eye Street, N.W., Suite 506
Washington, D.C. 20006

* Steven C. Sholly
304 South Market Street
Mechanicsburg, Pennsylvania 17055

* Daniel M. Pell, Esquire
ANGRY
32 South Beaver Street
York, Pennsylvania 17401

* Gail Bradford
ANGRY
245 West Philadelphia Street
York, Pennsylvania 17404

William S. Jordan, III, Esquire
Attorney for People Against Nuclear Energy
Harmon & Weiss
1725 Eye Street, N.W., Suite 506
Washington, D.C. 20006

* Robert Q. Pollard
609 Montpelier Street
Baltimore, Maryland 21218

* Chauncey Kepford
Judith H. Johnsrud
Environmental Coalition on Nuclear Power
433 Orlando Avenue
State College, Pennsylvania 16801

* Marvin I. Lewis
6504 Bradford Terrace
Philadelphia, Pennsylvania 19149

* Marjorie M. Aamodt
R. D. 5
Coatesville, Pennsylvania 19320